Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/P/356
USSR
May 1964
Original: RUSSIAN

Confidential until official release during Conference

NEUTRON SLOWING DOWN IN HYDROGENEOUS MEDIA

L.N. Yurova, A.A. Poljakov, S.B. Stepanov, G.A. Alimov

Fast neutron slowing down and the process of thermalization in hydrogeneous media were thoroughly studied in many papers. But still it is difficult to say that problem has been studied to the bottom. For instance, numerous experimental results of investigations of the neutron slowing down in water-uranium media do not agree with the calculated results. There is limited information of the influence of the medium heterogeneity on the neutron slowing down, the thermalization is not studied enough in various hydrogeneous media at high temperatures. That is why the new experimental information is of great scientific and practical interest for understanding the physics of the reactors with a hydrogen moderator.

This paper deals with the results of the experimental determination of two integral values in various media, lengths of fast neutron slowing down and diffusion coefficient of thermal neutrons.

I. FAST NEUTRON SLOWING DOWN

Experimental study of the fast neutron slowing down in non-multiplying media

Fast neutron slowing down was studied in hydrogeneous media: light water (H_2O) diphenyl ($C_{I2}H_{IO}$) isopropyldiphenyl ($C_{I5}H_{I6}$) as well as in the media ferrum + water; alluminium + water; ferrum + diphenyl and alluminium + diphenyl. Space distribution measurements of slowing down neutrons were carried out in the conditions of "limited medium". The term "limited medium" means the investigated medium of limited size surrounded by an infinite "reflector". Such geometry of the experiment gave the experimental distribution of the slowing down neutrons corresponding to the infinite media with limited quantity of investigated substances. A similar principle of measurement was applied to

the studying of physical parameters limited according to the size of multiplying media placed in the critical system.

A mathematical method published in the paper was used for a theoretical argument of using the experimental method and for the choice of the system size. / 3 /. As it was expected the results of the calculations proved that in the conditions of the cylindrical geometry of the investigated medium placed into the infinite "reflector" the diameter of the cylinder must be not less than four lengths of slowing down. ($4\ L_s$).

Cylindrical geometry with 4I cm. in diameter and II0 cm. high placed into the infinite graphite "reflector" was used for all experimental investigations of the neutron slowing down in non-multiplying media. In order to confirm the given geometry experimentally space distribution measurements of neutron (Po + Be) source ware made in the following media: diphenyl + graphite "reflector" and diphenyl + diphenyl "reflector". The measured density distributions of slowing down neutrons up to indium resonance in this conditions appear to be identical within experimental errors. The values of the square length of slowing down calculated according to this distributions are correspondingly equal to: (I02,3 $\pm$ 3,5) cm^2 - for the diphenyl + graphite medium and (I02,5 $\pm$ 4,I) cm^2 - for the diphenyl + diphenyl medium.

Investigating fast - neutron slowing down in metal-hydrogeneous media the experimental systems consisted of a moderator with evenly distributed metal rods. Two groups of media were studied: quasihomogeneous media in which metal rods were of I0 mm diameter and heterogeneous media with 40 mm rods.

Measurements were made for neutrons of fission spectrum U-235 and for neutrons of (Po + Be) source. Neutron fission source was a convertor made of metallic highly enriched uranium placed into a thermalizing column of a heavy water reactor of the Academy of Sciences of the USSR.

The space distributions were measured by 90 mg/cm^2 thick indium foil, the sizes of which were chosen in order to preserve the "point" sizes of the source and detectors. Corrections were introduced into the calculated values of the foil activities.

These corrections conserned: I. the contribution of the neutrons with energies above the first resonance level of indium, and 2. the corrections concerning the thickness of the neutron fission source. All the measured distributions were calculated to 0,3 mm /5/ thickness of the source. Square length of slowing down were calculated from the following equation:

$$L_s^2 = \frac{1}{6} \cdot \frac{\int_0^{r_0} A(r) r^4 dr + \int_{r_0}^{\infty} \frac{K e^{-r/\lambda}}{r^2} r^4 dr}{\int_0^{r_0} A(r) r^2 dr + \int_{r_0}^{\infty} \frac{K e^{-r/\lambda}}{r^2} r^2 dr} \qquad (1)$$

where $A(r)$ - indium foil activity;

K, λ - constants measured in the experimental neutron distribution for an region distant of the source where a linear law is present $\ln(Ar^2) = f(r)$

The results of the square length of slowing down values are given in Tables I, II and Figures I, 2.

The experimental study of the Neutron slowing down in a critical system

In the experimental study of neutron slowing down in a critical system was studied in the water-uranium and uranium - isopropyl diphenyl media. Due to neutron leakage measurements, corresponding to various geometries of the core, it was possible to determine neutron migration areas /6/. The periods of the doubled power system were measured in the experiment with the change of the core critical height. According to these periods the values of the reactivity were calculated using the formula of inhour. /7/. The contribution of delayed neutrons of various energetic groups into measured values of reactivity were taken into account.

β_i values were used in the "inhour" formula instead of the constant outputs of the delayed neutrons. These values β_i^{ef} were determined as follows:

$$\beta_i^{ef} = \frac{1 + \varkappa^2 L_s^{2pr}}{1 + \varkappa^2 L_{si}^{2de'}} \qquad (2)$$

where L_s^{2pr} - square length of slowing down of prompt neutron fission;

$L_{si}^{2de'}$ - square length of slowing down of the delayed neutron "i" group;

$\varkappa^2$ - buckling.

From $\frac{M^2}{K_\infty}$ the given values of the reactivity system the relations were determined according to the following equation:

$$\frac{\partial \rho}{\partial h} = - \frac{2\pi^2}{H^3 K_\infty} M^2 \qquad (3)$$

where H - critical height;
h - the exceeding of the critical height;
K_∞ - neutron multiplication coefficient for the infinite media.

The necessary buckling values for M^2 calculations were obtained from the critical experiments from the measurements of the axial and radial density neutron distributions in the core. These distributions were measured with copper foils.

The experimental plant was a zero power reactor, the core of which was located in a hexagonal tank made of stainless steel, its side being 50 cm. The core was surrounded by an infinite side reflector made of the moderator; there was no upper buttend reflector, the bottom of the tank I5 mm being the lower buttend reflector. Fuel elements were hollow cylinders of stainless steel filled with U_3O_8 (the size of the inner tube being 9,0 x 0,4 mm, of the outer tube - I3,4 x 0,2 mm). The fuel elements were located in alluminium spacing lattices. Due to lattice pitch variations it was possible to investigate the neutron slowing down in the media with different relationship of fuel and liquid moderator volumes.

In this experiment square neutron migration areas values were obtained. This values did not differ from the within the accuracy of the experiment, the diffusion length values of neutrons being negligible small in given multiplying media.

The value L_s^2 (thermal) is given in Table I

The calculation of square length of slowing down by the momentum method in multigroup approximation

Neutron transfer equation in multigroup approximation with elastic and non-elastic slowing down may be given as follows /8/:

$$\frac{\partial \varphi_1^j}{\partial z} + \sum_d^j \varphi_o^j = \sum_{e=1}^{j-1} \sum_{in}^{e \to j} \varphi_o^e + \alpha_o^j \sum_{e=1}^{j-1} \beta_o^e \varphi_o^e + \left(\frac{\zeta \Sigma_s}{\Delta u}\right)^{j-1} \varphi_o^{j-1} + x^j \delta(z) + x^j \sum_e^e \nu \sum_f^e \varphi_o^e \qquad (4)$$

$$\frac{1}{3} \frac{\partial \varphi_o^j}{\partial z} + \sum_1^j \varphi_1^j = \alpha_1^j \sum_{e=1}^{j-1} \beta_1^e \varphi_1^e$$

where $\sum_d^{e \to j}$ - is group scattering cross section descent from "j" group;

$\sum_{in}^{e \to j}$ - is group inelastic scattering cross section from "e" group into "j" group;

$\left(\frac{\zeta \Sigma_s}{\Delta u}\right)$ - is group sections of neutron slowing down by nuclei with A > I;

x_e^j - source functions;

$x^j \sum_e^e \nu \sum_f^e \varphi_o^e$ - is intergal neutron flux and current within the group;

$\varphi_o^j \varphi_1^j$ - takes into account the neutron multiplication in epithermal energy region.

$$\alpha_o^j = \left(1 - e^{\Delta u_j}\right) e^{-u_{j-1}}$$

$$\beta_o^e = \frac{1 - e^{-\Delta e}}{\Delta u e} e^{u e}$$

$$\alpha_1^j = \frac{2}{3}\left(1 - e^{-3/2 \Delta u_j}\right) e^{-3/2 u_{j-1}}$$

$$\beta_1^j = \frac{2}{3} \frac{1 - e^{-3/2 \Delta u e}}{\Delta u e} e^{3/2 u e}$$

Resonance effects in expression (I) are taken into account by adding the following value to leakage sections:

$$\delta \sum_o^j = \sum_{i=j-1}^{j} \rho \frac{J_{эф} \phi}{\Delta u_j} \qquad (5)$$

356

If one adds Fourier transformation to this system (1) and then make expansion "P", we shall obtain a system of relative equations for the square length of slowing down determination:

$$\sum_{\alpha}^{j} \varphi_{00}^{j} = \sum_{e=1}^{j-1} \sum_{i=1}^{e+j} \varphi_{00}^{e} + \alpha_0^{j} \sum_{e=1}^{j-1} \beta_0^{e} \varphi_{00}^{e} + \left(\frac{\Sigma_s'}{\Delta u}\right)^{j-1} \varphi_{00}^{j-1} + x^{j} \sum_{e} \nu \sum_{f}^{e} \varphi_{00}^{e} + x^{j}$$

$$\sum_{1}^{j} \varphi_{11}^{j} = \frac{1}{3} \varphi_{00}^{j} + \alpha_1^{j} \sum_{e=1}^{j} \beta_1^{e} \varphi_{11}^{e} \qquad (6)$$

$$\sum_{\alpha} \varphi_{00}^{j} = 2\varphi_{11}^{j} + \sum_{e=1}^{j-1} \sum_{i=1}^{e+j} \varphi_{02}^{e} + \alpha_0^{j} \sum_{e=1}^{j-1} \beta_0^{e} \varphi_{02}^{e} + \left(\frac{\Sigma_s'}{\Delta u}\right)^{j-1} \varphi_{02}^{j-1} + x^{j} \sum_{e} \nu \sum_{f}^{e} \varphi_{02}^{e}$$

$$L_s^{2j} = \frac{\varphi_{02}^{j}}{2\varphi_{00}^{j}}$$

In the expression (3) φ_{ne}^{j} are space angular moments of the neutron distribution function of "j" group in the infinite media with the plane isotropic source.

A program of numerical calculation L_s^2 is made for a computor /8/. The calculated values of square length of the slowing-down in hydrogeneous media are given in Tables I,II and Fig.1.

Measurement Results

"Quasi-homogeneous media". The results of the measurements and calculations L_s^2 for pure moderators and quasi-homogeneous metal–water media are given in Table I. There is good agreement between experimental and calculated data L_s^2 which proves the valuability of the calculation method of the neutron slowing down as well as multigroup constant systems used in calculations for the discription of the neutron slowing down process in hydrogeneous media.

Heterogeneous Media. The results of the neutron moderation measurements are given in Table II and Fig. I and 2. There is great difference in values L_s^2 with the same concentration of metal in water. In all cases the measuremented values L_s^2 for heterogeneous media exceeded the corresponding values for quasi - homogeneous media. The same is true for diphenyl - metal media for which the measured values L_s^2 are much greater than the calculated ones, obtained by the method applied to homogeneous media. This difference can be explained by a sharply expressed anisotropy of neutron diffusion in heterogeneous media.

In the previous papers dealing with fast neutron slowing down in metal-water media was not observed such sharp anisotropy, as the investigated media were, actually, quasi-homogeneous /9/.

The results obtained are evidently of interest for physical calculations of reactors, the core of which is sharply heterogeneous.

II. THERMAL NEUTRON DIFFUSION

Experimental Methods

Neutron diffusion in hydrocarbons (cyclic and non-cyclic compounds) has been studied in the medium with a wide range of temperatures (from melting to boiling). For investigation purposes the method of pulsed neutron source /II/ has been used.

Thermal neutron distribution in the medium may be described by means of a one-group diffusion equation in which diffusion constants averaged by the thermal neutrons spectrum are used:

$$\overline{D(v)} \cdot \Delta n - \overline{\Sigma_a \cdot v} \cdot n = \frac{\partial n}{\partial t} \qquad (7)$$

where $\overline{D(v)} = \frac{\overline{\lambda_{tr}(v) \cdot v}}{3}$ - diffusion coefficient in the infinite medium;

$\overline{\Sigma_a \cdot v}$ - neutron absorption rate in the infinite medium.

For a finite medium the changing of thermal neutron density in the course of time may be described as a sum of exponentially attenuating harmonics. A sufficient period of time having passed, all harmonics attenuate, except the main one, and neutron distribution is described by the law $e^{-\alpha t}$, where decay constant

$$\alpha = \overline{\Sigma_a \cdot v} + \overline{D(v)} \cdot \mathcal{B}^2 - (C_D - C_T) \cdot \mathcal{B}^2 \qquad (8)$$

$\mathcal{R}$ – geometrical buckling;

C_D – diffusion cooling coefficient;

C_T – non-diffusion correction.

Description of Experiment

Measurements were made with the cyclotron of the Academy of Science of the USSR. The unit consisted of a cylindrical tank 30 cm in diameter, shielded with a B_4C layer of 10 mm thickness and with a 0,5 mm cadimium layer. Into the tank there was inserted a cadmium "piston" by means of which the geometrical dimensions of the medium under investigation could be changed quickly and easily. Under the tank bottom a block of counters BF_3 was placed separated and shielded from the tank bottom with the help of a cadmium "mask" the shape of which can be determined by the function $\tau J_0(2{,}405 \frac{r}{R})$.

This "mask" decreased the influence of high harmonics of the neutron density for high geometrical buckling. For heating the medium an electric furnace was placed on the side surface of the tank. For temperature control a thermocouple was placed inside the tank, the accuracy of measuring the temperature being $\pm 2°C$.

The calculation of the unit and the control tests were carried out with water at the temperature of 21°C. For determining the effect of mutual arrangement of the cyclotron target and the unit, measurements were at different distances from and different angles to the target. The results remained the same within the experimental error. As water shielding and other scalterers were close to the unit, but the target was rather far from it, the fast neutron source was practically volumetric.

The liquid was heated to high temperatures and therefore the minimum distance between a block of counters and the tank bottom could be 2 cm. In connection with this all possible distortions of the decay constant were measured because neutrons of various energies covered the distance from the tank bottom to the conters in different time intervals.

The results of the experiment showing the effect of the covered distance on the decay constant shows that this effect in negligible from $\mathcal{R} = 0{,}240$ cm^2. The measurements were for $\mathcal{R}$ which were not larger than this quantity.

The cyclotron had a pulse-repetition rate of 500 cycles and the pulse duration of 6–8 μsec.

Time analysis of the decay of thermal neutron flux was made with the 60-channel time analyzer, the triggering of which

was synchronous with the cyclotron start-up. The delay of the time-analyzer triggering relative to the fast neutron pulse varied with-in 100-300 μ sec. The time channel width was 2-8 μ sec. The vibration of the base was not more than 0,25 μ sec.

The method of treatment the results

The values α, $\overline{\Sigma_a \cdot v}$, $\overline{D(v)}$, $(C_D - C_T)$ were calculated by the method of least squares. The dependance of the diffusion coefficient from the temperature of the medium for all the investigated materials is well discribed by the law:

$$\frac{\rho}{\rho_0} \overline{D(v)}_T = A \cdot T^{\frac{a}{2}} \qquad (9)$$

where $\overline{D(v)}_T$ - is the diffusion coefficient, averaged by the spectre of thermal neutrons at the temperature of medium T;

ρ, ρ_0 - is the density of the medium at given temperatures T, To;

a - is the parameter, depending upon the molecule structure of the material.

The results of the treatment of experimental data are given in table III.

It is natural to suppose, that the established neutron spectrum in an infinite medium oveys to the Maxwell distribution:

$$W(v)dv = A_0 e^{-v^2/v_0^2} v^2 dv$$

where $v_0 = \sqrt{\frac{2KT}{m}}$ -is the most probable velocity of the neutrons.

Strickly speaking, the spectre of the thermal neutrons obeys to the Maxwell distribution only in nonabsorbing media. But as the experiment shows one can consider the spectre as a Maxwell one if the absorption is weak. This allows to define the value v_0 within a small mistake. In this case one uses the medium temperature as a parameter.

Then the equation /9/ may be written as follows:

$$\frac{\rho}{\rho_0} \overline{D(v)}_T = B v_{0T}^a \qquad (10)$$

The diffusion coefficient averaged by the Maxwell spectre can be written in the form of:

$$\overline{D(v)} = \frac{\int_0^\infty e^{-v^2/v_0^2} v^2 D(v) dv}{\int_0^\infty e^{-v^2/v_0^2} v^2 dv} \qquad (11)$$

From the equation (10) and (11) one can get the ratio (11) for defining the differential value of the diffusion coefficient of neutrons with the velocity v by the values of the diffusion coefficients of neutrons for the most probable velocity v_0

$$D(v) = D(v_0)\left(\frac{v}{v_0}\right)^a \tag{12}$$

$$D(v_0) = \frac{B}{m(a)}\, v_0^{\,a} \tag{13}$$

where $m(a)$ - is the coefficient of the averaged function, depending upon the kind of neutron spectrum and of the index a

The averaged diffusion coefficient of neutrons by the Maxwell spectrum can be defined by the differential value of the diffusion coefficient of neutrons for the most probable velocity.

$$\overline{D(v)} = D(v_0)\frac{\int_0^\infty e^{-v^2/v_0^2} v^{2+a}\, dv}{v_0^a \int_0^\infty e^{-v^2/v_0^2} v^2\, dv} = m(a)\, D(v_0) \tag{14}$$

From the equation (13) one can get the dependance of the transport length upon the neutron velocity

$$\lambda_{tr}(v_0) = \frac{3B}{m(a)}\, v_0^{\,a-1} \tag{15}$$

and we finally have

$$\overline{D(v)}_T = m(a)\cdot D(v_0)\cdot\left(\frac{v_T}{v_0}\right)^a \tag{16}$$

$$\overline{\lambda_{tr}(v)}_T = m(a-1)\cdot \lambda_{tr}(v_0)\left(\frac{v_T}{v_0}\right)^{a-1} \tag{16$_1$}$$

It follows from the abovementioned, that the value of the dependance of the integral value $\overline{D(v)}$ upon the medium temperature and the knowledge of the kind of the spectrum of the thermal neutrons allows to get the differential value $D(v_0)$ from the equation (14).

And the differential value $\lambda_{tr}(v_0)$ from the equation

$$\lambda_{tr}(v_0) = \frac{3\overline{D(v)}}{m(a)\cdot v_0} \tag{17}$$

From the equations /16/ and /16$_1$/ it follows

$$\overline{D(v)} = \frac{2}{\sqrt{\pi}}\cdot\frac{m(a)}{m(a-1)}\cdot\frac{\overline{\lambda_{tr}(v)}\cdot\overline{v}}{3} \tag{18}$$

Therefore it is necessary to note in references (e.g.11,13) an in accurate ratio of the diffusion coefficient is used

$$\overline{D(v)} = \frac{\overline{\lambda_{tr}(v)} \cdot \overline{v}}{3} \qquad (19)$$

Got from the equation (19) has no definite physical sense. This concerns also to the definition of $\overline{\lambda_{tr}(v)}$ received from the experiments with stationary diffusion. In this case $\overline{L^2(v)}$ is an average of the spectrum of thermal neutrons:

$$\overline{L^2(v)} = \overline{T \cdot D(v)} = T \cdot \overline{D(v)} = m(a) \cdot T \cdot D(v_0) = T \cdot \frac{2}{\sqrt{\pi}} \frac{m(a)}{m(a-1)} \cdot \frac{\overline{\lambda_{tr}(v)} \cdot \overline{v}}{3} \qquad (20)$$

Then in order to get an information of the transport length value, it is necessary to find the value of $m(a)$. This can be done if one defines $\overline{L^2(v)}$ at different temperatures of the medium and the results are treated according the abovementioned method.

Results

From the values of the transport length one can calculate the transport cross section of the neutron scattering by the molecules of the medium, averaged by the neutron spectrum; it is also possible to obtain the differential value of the transport cross section of moleculs neutrons of the medium in the case of the Maxwell distribution. The latter makes it possible to calculate the transport cross section of one hydrogen atom, considering the atoms of carbon being free.

Thus it is possible to determine both the integral value of the transport cross section of the neutron scattering by the bound hydrogen and the differential cross section. The latter values must correspond to the values of the transport cross section obtained from the differential experiments.

In order to verify the methods of treatment of the experimental results, a comparison of the values of transport cross section of neutron scattering by the medium molecules was made. These values were obtained both from integral and differential experiments. The comparison was made for water. The results of the integral experiment for water /11/ were treated according to the abovementioned method /equation 17/.

From differential experiments on the scattering of neutrons with different energies with water $\cos\theta$ was taken from /12/ and σ_s - from /13/, and σ_{tr} was calculated. σ_{tr} was also obtained from /11/ using the expression (17). The results

agreed perfectly. Besides, from /11(3)/ σ_{tr} were obtained using (19). These values of σ_{tr} differ greatly from the results of the differential experiment. The difference is of about 40% (fig.3).

Table III gives the values of the parameter of a and σ_{tr}^{H} by bound hydrogen for the investigated matter at the medium temperature 18°C, including the data of water and dowtherm /11,14/ treated according to the abovementioned method.

Transport cross section of the bound hydrogen σ_{tr}^{H} changes from 33 barns for $a \sim 1$ to 100 barns for $a \sim 3$. There is a whole spectrum of intermediate values (fig.4) depending on the structures of compounds.

Transport cross section of the neutron scattering by the bound hydrogen strongly depends on the medium temperature in the case of non-cyclic compounds.

Acknowledgements the authors express their gratitude to S.A. Gavrilov, I.G. Morosov, E.I. Injutin, G.I. Sidorov, B.M. Stasevetch, V.V. Okorokov, V.V. Dolganov, V.A. Fedorovitch for their help, as well as V.P. Kotchergin, A.F. Zagirko for making calculations on the computor.

References

/1/ 1./ H.GOLDSTEIN, P.F. ZWEIFEL, D.G. FOSTER, Jr.
The second United Nations international conference on the peaceful uses of atomic energy. Report 2375.

2./ H. KOUTS, R.SHER, S.R. BROWN, D.KLEIN, S.STEIN, R.L.HELLENS, H.ARNOLD, R.M.BALL, P.W.DAVISON
The second United Nations international conference on the peaceful uses of atomic energy. Report 1841.

3./ A.W.McREYNOLDS, M.S.NELKIN, M.N.ROSENBLUTH, W.L.WHITTEMORE.
The second United Nations international conference on the peaceful uses of atomic energy. Report 1540.

4./ C.D.JOANOU, A.J.GOODJOHN, N.F.WIKNER.
"Nuclear Sci. and Eng.B., 171-189/1962/.

/2/ G.A.STOLYAROV, Z.B.KOMISSAROV, V.P.KATKOV, U.V.NIKOLSKY.
The meeting of the Academy of Science of the USSR on the peaceful use of atomic energy 1955. A of Sc. of the USSR /1955/.

/3/ A.E.GLAUBERMAN, I.I.TALJANSKY.
"Atomic energy" N 3 /july 1957 /.

/4/ L.N.Jurova, A.A.POLJAKOV, S.B.STEPANOV, V.B.FROJANSKY.

"Neutron physics" Gosatom isdat /19603.

/5/ 1./ L.N.JUROVA,A.A.POLJAKOV,A.A.IGNATOV.
"Atomic Energy" v.12 part 2 /1962/.

2./ L.H.JUROVA,A.A.POLJAKOV,A.A.IGNATOV.
"Some questions of engineering physics" Gosatomisdat /1963/.

/6/ S.KRASIK, A.RADKOWSKY
International conference on peaceful use of atomic energy.Report F/601 /1955/.

/7/ P.GLASSTONE and M.EDLUND.
The principals of the theory of nuclear reactors. Foreign literature publishing house.M.1954.

/8/ 1./ G.I.MARCHUK.
"Methods of calculation of the nuclear reactors state publish house of atomic science and technique .M.1961.

2./ V.P.KOCHERGIN,A.F.ZAGIRKO.
"The program of the calculation of the square of the neutron slowing down length in homogenious media in multigrouped approximation.
The phys.energ.Inst.of the State Comittee of the Sov. of Minist of the USSR Report N / 557

/9/ 1./ A.MUNN,A.B.PONTECORVO.
"Car.J.Res.".A.25 157-167 /1947/.

2./ M.REIER, F.OBENSHAINE and R.L.HELLENS.
"Nuclear Sci.a.Eng.4 ,1-11 /1958/.

/10/ C.D.JOANOU, A.J.GOODJOHN, N.F.WIKNER.
"Nuclear Sci.and Eng.13,171-189,/1962/.

/11/ 1./ A.V.ANTONOV etal.
The first international conference on peaceful use of atomic energy.Report P/661.

2./ Ct.F.von DARDTEL.
Frans.Royal Inst.Techn.N75 STOKHOLM /1954/

3./ A.V.ANTONOV et.al.
"Atomic Energy" v.12 part 1 /1962/

4./ G.F.von Dardel and N.G.SJÖSTRAND
Phys.Rev. 96, 1245/1954/.

/12/ V.I.MOSTOVOY et.al.Report for the Geneva conference, 1964.

/13/ Donald J.HUGHES and ROBERT B.SCHWARTZ,
"Neutron cross section" BROOKHAVEN national laboratory Jan.1 /1957/.

356

/14/ M.KUCHLE "Nucleonic" 2 131 /1960/.

TAB.1

Media	Neutron source	V_u/V_{mod}	Ls^2 /cm/^{2}			
			exper.		calcul.	
			Ls^2(1,46ev)	Ls^2(term)	Ls^2(1,46ev)	Ls^2 (term)
(H_2O)	Fission neutrons U-235	-	26,7±1,0	-	27,3	-
$(C_{12}H_{10})$		-	52,1±2,4	-	50,8	-
$(C_{15}H_{16})$		-	43,2±1,8	-	43,4	-
$C_{15}H_{16}+U_3O_8$		0,07	-	51,5±1,5	-	52,1
		0,09	-	54,4±1,6	-	54,4
		0,12	-	57,2±1,9	-	57,8
		0,17	-	64,0±1,7	-	63,2
		0,22	-	73,3±2,3	-	69,8
$H_2O+U_3O_8$		0,07	-	35,5±1,4	-	33,1
		0,09		37,2±1,5	-	34,7
		0,17		43,1±2,3	-	41,1
(H_2O)	(P_o+Be) source	-	56,7±0,9	-	58,5±1,5	-
$(C_{12}H_{10})$		-	102,3±3,5	-	99,1	-

TAB.II

Media	Neutron source	Vmet/Vmod		$L^2s(1,46ev)/cm^2$ exper.	$L^2s(1,46ev)/cm^2$ calcul.
(H_2O)	(Po+Be)source	-	-	56,7±0,9	58,5±1,5 /10/
H_2O+Fe		1:3	quisi.gom. media	55,6±1,0	-
			get.media	63,1±1,2	-
		2:3	quisi gom. media	53,1±1,4	-
			get.media	69,1±1,7	-
		1,42	quisi gom. media	62,6±1,7	-
			get.media	77,7±1,8	-
H_2O+Al		1:3	quisi gom. media	72,1±1,6	-
			get.media	79,6±2,1	-
$(C_{12}H_{10})$		-	-	102,3±3,5	99,1
$C_{12}H_{10}+Fe$	Fission neutrons U-235	1:3	get.media	75,0±4,4	68,5
$C_{12}H_{10}+Al$		1:4	- " -	87,8±2,3	79,9

TAB. III

Name	Index	t°C	$\overline{\Sigma_a v} \cdot 10^4$ sec	$\overline{D(v)} \cdot 10^4$ $cm^2 sec^{-1}$	$(C_a - C_r) 10^4$ $cm^4 sec^{-1}$	ρ $g.cm^{-3}$	a	$\sigma''_{tr}(2200)$ Å
Benzene	C_6H_6	14	0,280±0,041	5,37±0,51	0,86±0,51	0,885	0,99±0,28	33,1±0,7
		27	0,252±0,021	5,24±0,21	0,16±0,47	0,871		
		34	0,248±0,019	5,66±0,36	1,26±0,38	0,863		
		38	0,248±0,017	6,27±0,41	2,92±0,91	0,859		
		52	0,283±0,042	6,09±0,59	3,42±1,70	0,845		
		71	0,310±0,031	6,13±0,33	2,34±0,83	0,824		
Diphenyl	$C_{12}H_{10}$	123	0,280±0,011	7,70±0,08	8,20±0,50	0,955	1,35±0,05	32,5±0,3
		150	0,267±0,014	8,24±0,08	8,97±0,60	0,930		
		175	0,260±0,012	8,75±0,09	10,4±0,8	0,910		
		207	0,253±0,016	9,55±0,10	12,0±0,8	0,884		
		243	0,236±0,010	10,40±0,11	13,9±0,9	0,850		
Diphenylmetane	$(C_6H_5)_2CH_2$	34	0,390±0,021	4,05±0,21	1,12±0,36	0,994	1,61±0,24	49,0±3,0
		110	0,386±0,017	5,35±0,23	2,62±0,82	0,936		
		160	0,411±0,028	6,80±0,30	4,9±1,2	0,897		
		230	0,433±0,026	6,95±0,33	3,5±1,7	0,842		
Dowtherm [14]	$C_{12}H_{10}O_{mix}$						1,90±0,08	46,5±1,5
Water [11]	H_2O						1,91±0,02	44,0±0,08
Diphenylether	$(C_6H_5)_2O$	97	0,200±0,028	8,16±0,48	5,9±1,3	0,920	2,16±0,14	46,0±3,3
		110	0,243±0,019	9,87±0,61	7,6±2,0	0,908		
		175	0,230±0,017	10,48±0,84	8,6±1,9	0,845		
		204	0,200±0,030	13,52±1,14	13,9±2,9	0,816		
Gasoil		24	0,482±0,020	3,23±0,21	2,2±0,6	0,842	2,44±0,09	61,3±3,1
		69	0,465±0,030	3,95±0,27	2,9±0,5	0,813		
		144	0,430±0,030	5,62±0,42	5,1±1,3	0,762		
Isopropyldiphenyl	$(C_6H_5)_2C_3H_7$	18	0,433±0,021	3,14±0,04	1,24±0,08	1,000	2,58±0,08	82,1±1,2
		85	0,417±0,018	4,51±0,06	3,08±0,20	0,950		
		144	0,401±0,019	5,81±0,06	5,04±0,30	0,900		
		197	0,390±0,015	7,01±0,08	6,11±0,41	0,852		
		243	0,356±0,014	8,17±0,08	8,70±0,52	0,821		
Anisole	$C_6H_5OCH_3$	18	0,351±0,031	4,40±0,32	2,3±0,8	0,995	2,94±0,19	86,7±5,8
		80	0,332±0,041	5,83±0,44	5,7±1,1	0,935		
		150	0,311±0,052	7,57±0,40	7,8±1,9	0,865		
Tetradecane	$C_{14}H_{30}$	18	0,561±0,018	2,33±0,11	1,1±0,2	0,762	3,16±0,18	103,9±3,4
		76	0,533±0,011	3,19±0,16	2,0±0,4	0,724		
		160	0,487±0,016	5,33±0,21	4,2±1,0	0,668		
		248	0,349±0,013	7,89±0,22	6,8±2,0	0,596		

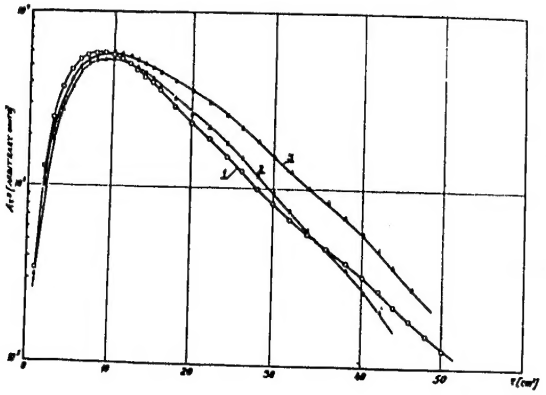

Fig.1. The distribution of the neutron flow of indium resonance in three media: 1-water; 2-"quasihomogeneous" medium $Fe+H_2O$ $V_{fe}:V_{H_2O}=2:3$; 3-heterogeneous medium $Fe+H_2O$ $V_{fe}:V_{H_2O}=2:3$.

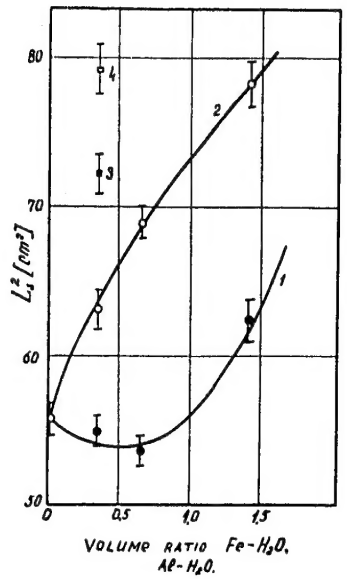

Fig.2. Square length of the neutron slowing down of the (Po+Be) source in metalic + water media: 1-"quasihomogeneous" medium $Fe+H_2O$; 2-heterogeneous medium $Fe+H_2O$; 3-"quasihomogeneous" medium $Al+H_2O$; 4-heterogeneous medium $Al+H_2O$.

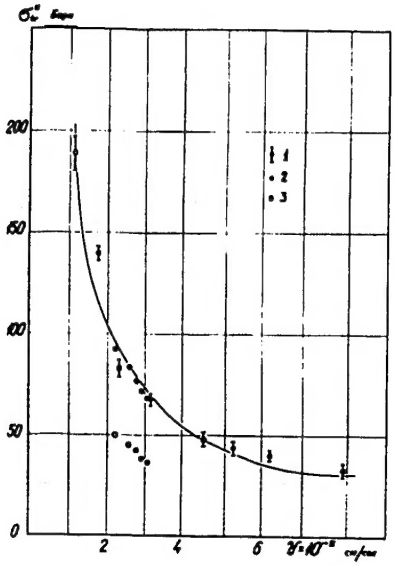

Fig.3. Transport cross section of the scattering by water, received by different methods. 1-differential experiment; 2-integral experiment received from the equation (17); 3-integral experiment received from the equation (19).

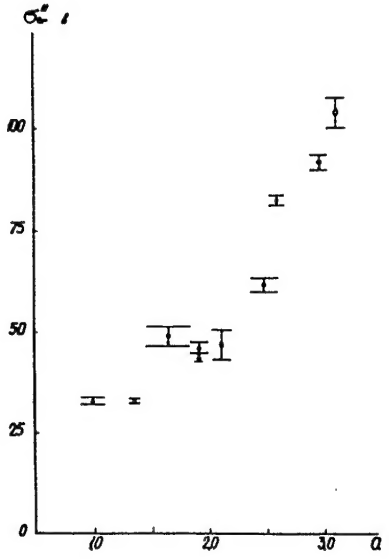

Fig.4. The dependence of σ_{tr} upon the parameter a